

DTIC FILE COPY

Best Available Copy

AD-A220 111

20030210206



MONTE CARLO DETERMINATION OF
GAMMA RAY EXPOSURE FROM A
HOMOGENEOUS GROUND PLANE

THESIS

Michael R. Howard
Captain, USAF

AFIT/GNE/ENP/90M-2

DISTRIBUTION STATEMENT A

Approved for public release;
Distribution Unlimited

DEPARTMENT OF THE AIR FORCE
AIR UNIVERSITY

AIR FORCE INSTITUTE OF TECHNOLOGY

Wright-Patterson Air Force Base, Ohio

DTIC
ELECTE
APR 05 1990

SC E D

00 00 05 117

AFIT/GNE/ENP/90M-2

MONTE CARLO DETERMINATION OF
GAMMA RAY EXPOSURE FROM A
HOMOGENEOUS GROUND PLANE

THESIS

Michael R. Howard
Captain, USAF

AFIT/GNE/ENP/90M-2

Approved for public release; distribution unlimited

MONTE CARLO DETERMINATION OF GAMMA RAY EXPOSURE
FROM A HOMOGENEOUS GROUND PLANE SOURCE

THESIS

Presented to the Faculty of the School of Engineering
of the Air Force Institute of Technology
Air University
In Partial Fulfillment of the
Requirements for the Degree of
Master of Science in Nuclear Engineering

Michael R. Howard, B.S.
Captain, USAF

March 1990

Accession For	
NTIS GRA&I	<input checked="checked" type="checkbox"/>
DTIC TAB	<input checked="checked" type="checkbox"/>
Unannounced	<input type="checkbox"/>
Justification	
By _____	
Distribution/	
Availability Codes	
Dist	Avail and/or Special
A-1	

Approved for public release; distribution unlimited

PREFACE

The purpose of this study was to determine the Source Normalization Constant caused from the fission fragments of three fuels, U-235, U-238, and Pu-239. The contribution from scattered photons was determined by using a Monte Carlo code, MORSE. These results were compared to other approximate methods previously performed. Another objective was to obtain a transport function which could be used with any other source and response function to calculate a dose from a ground plane source.

I would like to acknowledge several people who were instrumental in the successful completion of this study. The patience and guidance of Dr. Charles J. Bridgman made this study a success while allowing it to be a learning experience. Likewise, I would like to acknowledge LT CMDR Kirk Mathews for his insight on the Monte Carlo process. The guidance of Major Denis Beller was also appreciated. Finally, I would like to thank my brother Scott for his encouragement and tolerance.

Table of Contents

	Page
Acknowledgement	ii
List of Figures	v
List of Tables	vi
Abstract	vii
I Introduction	1
Background:	1
Purpose and Scope:	3
II Theory	5
Dose Rate:	5
Build-Up Factor:	7
Successive Scatters:	8
Monte Carlo:	9
Monte Carlo Technique:	9
Survival Rates:	13
Russian Roulette:	13
Splitting Technique:	14
Transport Function:	15
III Method of Analysis	18
MORSE Model:	18
Dose Rate:	21
Transport Matrix:	23
IV Results and Discussion	25
V Conclusions and Recommendations:	34
Appendix A: MORSE User Written Subroutines	37
Appendix B: MORSE Sample Input and Output	43
Appendix C: Sample Calculations	46
Appendix D: Gamma Emission Rates for U-235 and Pu-239	48
Appendix E: Transport Matrix	55
Appendix F: Mono-Energetic Source Conversion Coefficient	59

Appendix G: Source Normalization Constant verses Time	60
Bibliography	61
Vita	62

List of Figures

	Page
Figure 1 Geometry for Dose Calculations	5
Figure 2 Typical path traveled by a photon	10
Figure 3 Spherical coordinates	12
Figure 4 Mono-energy SCC, $E < 700$ kev	25
Figure 5 Mono-energy SCC $E > 700$ kev	26
Figure 6 Comparison of Mono-energy SCC	27
Figure 7 Comparison of Mono-energy SCC	29
Figure 8 Time dependent SCC	29
Figure 9 $K(t)$ vs Way-Wigner for U-235	31
Figure 10 $K(t)$ vs Way-Wigner for U-238	32
Figure 11 $K(t)$ vs Way-Wigner for Pu-239	32
Figure 12 Ratio of Calculated K to Way-Wigner Approximation	33

List of Tables

	Page
Table 1 Composition of Air and Soil	20
Table 2 Source Normalization Constant at (1 hour)	30

ABSTRACT

The purpose of this study was to evaluate the gamma ray dose rate from an infinite homogeneous ground plane source of fission products from U-235, U-238, and Pu-239. This evaluation was previously carried out using approximate methods to model the contribution to dose resulting from gamma scatter in the air and in the ground. In this study a Monte Carlo method was used to convert the ground source to a dose rate. This was accomplished by using the MORSE code. Two intermediate parameters which were used to determine the dose rate are the Source Conversion Coefficient (SCC) (rads-tissue/hr)/(MCl/km²) and Source Normalization Constant (K) (rads-tissue/hr)/(kt/km²). The SCC for mono-energetic gamma-rays was calculated from MORSE and then used to determine the time dependent SCC for a variety of spectra. Both the energy spectra and the gamma emission rates which were used in this study were determined by Millage. The time dependent SCC's and K values are presented for the three fuels. The K values at one hour after burst were 8064 (rads-tissue/hr)/(kt/km²) for U-235 fission products, 8467 (rads-tissue/hr)/(kt/km²) for U-238 fission products, and 7666 (rads-tissue/hr)/(kt/km²) for Pu-239 fission products. In addition to these results, the MORSE code was also used to compute a transport matrix which will permit the use of any other source (fuel) spectrum and any other response function (rads-silicon, etc) to be converted to a ground dose rate. This transport matrix is also presented.

MONTE CARLO DETERMINATION OF GAMMA-RAY DOSE

FROM A HOMOGENEOUS GROUND PLANE SOURCE

I Introduction

Background:

Radioactive contamination of the ground can be caused from the fallout of a nuclear detonation or a nuclear accident. This contamination can cause serious complications for people and equipment. The dose rate is a measure used to determine the degree of damage. Calculations used to determine the dose rate from ground contamination are usually based on several assumptions which can cause variations in the answer. One of these assumptions is the impact of gamma ray scatter (in the air and ground) on the dose rate. There are various techniques used to determine the contribution of scatter to dose rates such as the Build-Up Factor method, the use of Successive Scatter, or Monte Carlo evaluation.

Glasstone and Dolan's The Effects of Nuclear Weapons, a commonly used reference for nuclear weapon studies, presents dose rate calculations which include the contribution of scattered photons. However, no discussion of the method used to treat scatter is presented by Glasstone and Dolan.

There is also no discussion on the different gamma-ray spectra generated by various fuel types. Glasstone and Dolan

do state the average photon energy of the source one hour after detonation is 700 kev (6:454). In reality, the average energy of the gamma-ray source is dependent on fuel type and time after burst.

In order to account for the various types of fissionable material in nuclear warheads, along with their fission product decay characteristics, Millage utilized two isotope generation and depletion codes, ORIGIN2 and DKPOWR (9:6). These codes determined the activities, gamma emission rates, and energy spectra at various times after detonation for the three major fuels. In his study, Millage used the build-up factor and successive scatter techniques to determine the effects of scatter on exposure rates. However, different assumptions were used within each technique. The build-up factor method assumed infinite homogeneous air while the successive scatter technique assumed the ground to be an absolute absorber. For the final results, Millage chose the successive scatter method because he believed the technique to be more accurate than the BUF method (9:9-15,34).

MORSE, a Monte Carlo transport code, will be used in this study to find the effects of scatter on the dose rate. The Monte Carlo results will be used to benchmark Millage's build-up factor results and his successive scatter results. The Monte Carlo results will also be used to establish a transport matrix to convert any gamma-ray spectrum and response

function to a dose rate. The Multigroup Oak Ridge Stochastic Experiment code (MORSE) is a general purpose Monte Carlo multi-group neutron and gamma-ray transport computer code. The version of the code, denoted as CCC-203/MORSE-CG was used. It is available from the Radiation Shielding Information Center (RSIC) located at Oak Ridge National Laboratory (5). The computer code, which will be referred to as MORSE in the remainder of this document, was run on DEC VMS computer systems.

Purpose and Scope:

The following goals were established for this study:

- (1) evaluation of the dose rate from a homogeneous ground plane source as determined by the fission products of three major fuels, U-235, U-238, and Pu-239,
- (2) examination of the major assumptions made by Millage and their effect on the dose rate,
- (3) evaluation of a transport matrix to convert any given gamma-ray spectrum and response function to a ground dose rate.

The gamma-ray spectra and gamma-emission rates used for the three fuels, U-235, U-238, and Pu-239, are those determined by Millage. The dominant limitations of this study are those imposed by Millage in his calculations of the source spectra and emission rates. The assumptions and limitations which

Millage used in determination of the spectra are listed below.

(1) No induced activity from the weapon itself. This could cause significant differences in both the energy spectrum and gamma emission rate depending on the weapon used.

(2) The fission products for U-235 and Pu-239 fuels were based on thermal fission while the fission products for U-238 are based on fast fission. The fission yield from weapon energy neutrons were ignored due to a lack of data and cross sections. However, Millage's results reveal that the neutron energy difference produces a negligible error (9:2-3).

Another assumption used in this study involves the gamma-ray source. The source is assumed to be distributed homogeneously over an infinite ground plane which is perfectly smooth. The dose rate for a rough ground plane would be less than that for an infinite smooth ground plane. This is due to the shielding provided by the terrain. A typical reduction factor used by Glasstone is 0.7 for moderately rough terrain. The actual reduction factor will depend on the actual terrain features. Other assumptions will be stated where applicable.

II Theory

Dose Rate:

The standard dose rate from a ground plane source is measured at a distance one meter above the plane (see Figure 1). One method used in the calculation of dose rates is the use of a parameter called the Source Conversion Coefficient (SCC). The SCC is the dose rate (Rad/hr) one meter above a ground plane source with an activity density of one Gamma-Megacurie per square kilometer. The term Gamma-Megacurie refers to the gamma emission rate of 3.7×10^{16} gammas per second (6:452-454). The SCC is a convenient parameter because it removes the gamma emission rate (GER) from the calculations. Only the gamma spectrum is used to calculate the Source Conversion Coefficient.

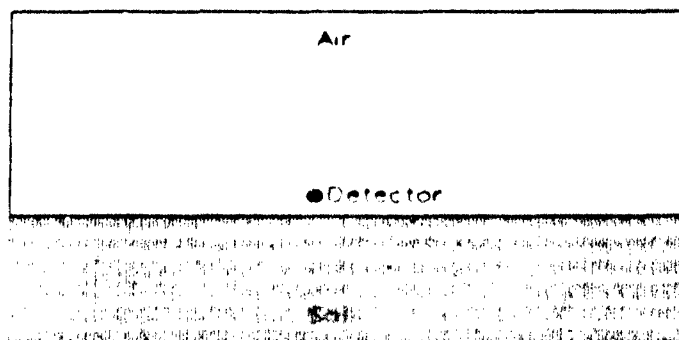


Figure 1
Geometry for Dose Calculations

Another term which is used in dose rate calculations is the Source Normalization Constant (K). The K value is the dose rate measured one meter above the ground containing the total activity released from a one kiloton weapon being homogeneously spread over one square kilometer (3:30). Although the Source Normalization "Constant" is a function of time, K(1hr) is often used in ground dose calculations. It can be determined by multiplying the Source Conversion Coefficient by the gamma emission rate one hour after detonation. The final dose rate can be determined by adjusting the K value by other factors such as: Way-Wigner approximation, fallout fraction, roughness factor, etc.

The dose is produced by photons which directly impinge on the detector and by photons which scatter to the detector. The photons which strike the detector before scattering are referred to as direct photons. Although the dose rate due to direct photons is relatively simple to calculate, the dose resulting from scattered photons is somewhat more complex. There are several methods which can be utilized to evaluate the scattered dose rate. Three of the techniques used are as follows: the Build-Up Factor method, the use of Successive Scatter, and Monte Carlo determination.

Build-Up Factor: Numerous studies have been made in the determination of build-up factors (BUF). Particularly, Bigelow and Kalansky both used the method of moments to calculate build-up factors. Bigelow determined the BUF for energies ranging from 100 keV to 10 MeV (2:58-100), while Kalansky determined the BUF energies below 1 MeV (7:140-141). The build-up factor is a multiplication factor used to determining the amount of energy which is deposited on a target by both direct and scattered photons compared to that deposited by direct photons alone. There are two major problems with this method of determining the dose rate.

The first problem is the fact that the method of moments requires the assumption of infinite homogeneous air which does not accurately describe the build-up from a ground plane source. The assumption of infinite homogeneous air will tend to overestimate the real BUF from a ground plane source. This is explained by the different cross sections of air and soil. The cross sections vary as a result of the various elements which compose air and soil. The higher the atomic number of an element the higher its absorption cross section will be due to photoelectric effect. Since there are lighter elements in air than in soil, a photon will lose more energy or vanish completely much quicker in soil than photons which travel through air. Thus, the BUF for a ground plane source will be lower than that of homogeneous air.

Another cause of error in the build-up factor method is the fact the BUF is an energy BUF. An energy BUF is deceptive because it measures the total energy deposited at a target and not the discrete energy per photon which strikes the target. For example, if the BUF for a one MeV photon source, in which one direct photon arrives at the target for a given number of mean paths, is three, there is no method to determine how the remaining two MeV arrived at the target. There could have been four 500 keV photons or two 500 keV photons along with four 250 keV photons. The possibilities are limitless. Since the actual energies of the gamma-rays must be known in order to determine the true response of the detector, the BUF method cannot accurately determine the dose rate. Assuming that the scattered photons have the same response function as the direct photon will tend to under estimate the dose rate because the response function usually increases as the energy of the photon decreases.

Successive Scatters: In the method of successive scatter used by Millage in his study, this problem was modeled as an infinite homogeneous source distributed over a ground plane where the ground plane was an absolute absorber. Assuming that the ground is a pure absorber will underestimate the actual dose rate as a result of photons being lost which would normally be

scattered from the ground to the air. In order to determine the true dose rate, the photons in the ground must also be tracked.

Monte Carlo: The Monte Carlo method was used in this study in order to avoid making the assumptions which were made in the BUF and successive scatter calculations performed by Millage. The great strength of Monte Carlo technique is the degree of physical realism that can be retained in the model. Unlike the successive scatter method, which assumed the ground to be an absolute absorber, the Monte Carlo method tracks the photons in both the air and the ground. It also determines the discrete energies of the photons striking the target which will correct the inability of the BUF to accurately predict the energy dependent dose rate.

Monte Carlo Technique:

In the basic form of the Monte Carlo method, a game is carried out on a computer simulating the actual physical processes which govern the particle's behavior. The term Monte Carlo comes from the use of random numbers which determine the outcome of chance events. A life history of each photon leaving the source is recorded as it travels from collision

site to collision site. This is accomplished by using random sampling techniques which describe the probability laws that govern the photon's 'random walk' through a medium.

The term 'random walk' refers to the path a photon travels. A photon will travel in a zigzag manner shown in Figure 2 as a result of scattering events. The distance and direction traveled after an interaction is determined by the use of random numbers.

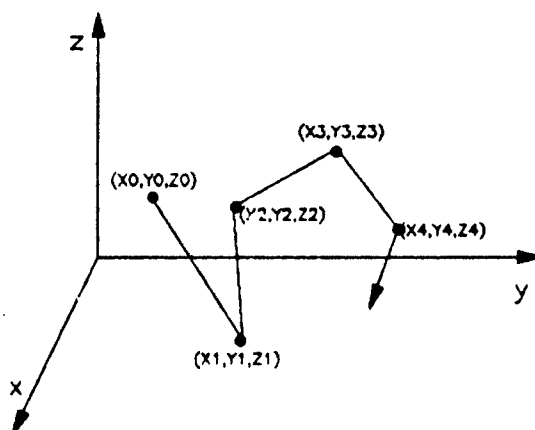


Figure 2
Typical path traveled by a photon in a medium

The path length traveled is determined by a random number described by the function in equation 1. It is derived from the probability that a photon will travel a distance s without having an interaction. This probability is given by equation 2.

$$s = -\frac{\ln(r)}{\mu} \quad (1)$$

$$P = e^{-\mu s} \quad (2)$$

where

P = Probability that a photon travels s without an interaction

s = distance traveled by photon (cm)

μ = total cross-section (cm^{-1})

r = random number ($0 < r < 1$)

The direction a particle travels is also determined by the use of random numbers. This is usually accomplished by randomly picking several numbers, n , such that $-1 \leq n \leq 1$. The random numbers are then used as the sine and cosine of the angles ϕ and θ . These are directional angles in the spherical coordinate system as seen in Figure 3. With the sine and cosine of the directional angles determined, combined with the distance which the particle will travel, the next collision site can be calculated with equations 3, 4, and 5.

$$X_{i+1} = X_i + S_i(\sin \theta_i \cos \phi_i) \quad (3)$$

$$Y_{i+1} = Y_i + S_i(\sin \theta_i \sin \phi_i) \quad (4)$$

$$Z_{i+1} = Z_i + S_i(\cos \theta_i) \quad (5)$$

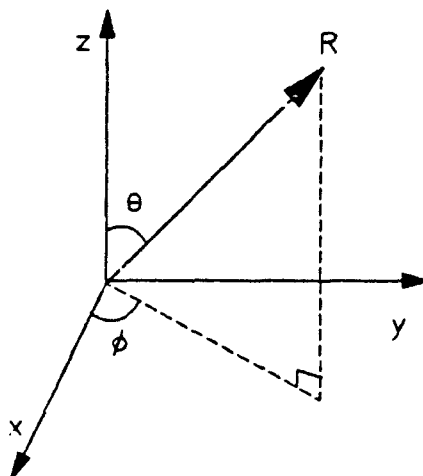


Figure 3
Spherical coordinate system

As the gamma-ray is traced, a record of its energy, direction, and position is maintained. A particle is tracked until it no longer contributes information of interest to the problem. The particle is then terminated and a new source particle is started and the process is repeated (12:274-277).

Since Monte Carlo is based on statistical concepts, the answer it gives is not unique. However, it is based on an estimate which lies within a confidence interval around the true answer. There are several variance reduction techniques which can be used. They include, but are not limited to, survival weights, Russian roulette, and splitting techniques.

Survival Rates: When the method of survival weights is introduced in a Monte Carlo calculation, all particles survive every collision. Therefore, it assumes that all interactions are scattering events and not absorption events. This would distort the calculations since it no longer represents the real physical processes which occurs. To counteract this result, a survival weight is given to each particle. The weight is adjusted after each interaction by multiplying the weight by a factor of $\mu_s(E)/\mu(E)$, where $\mu_s(E)$ is the scattering cross section and $\mu(E)$ is the total cross section for a photon. This factor is the probability that the interaction was a scattering event (12:292). Since this method has no killing mechanism, additional criteria must be used to terminate a particle's history. One method is to use a Russian Roulette technique.

Russian Roulette: Russian Roulette is the name given to the game played with a low weight particle to determine if it is necessary to continue its life history. After each scattering event, the particle's weight is tested to determine if it is below a previously stipulated weight value. If the weight is found to be less than the stated value, the game of Russian roulette is initiated. In this game, a particle has a previous fixed chance of survival, say a 1 in 10 chance. A random number between 0 and 1 is then selected. If the random number is greater than 0.1 the particle is killed and its history is

terminated; however, if the random number is less than 0.1 the particle survives and its history is continued. In order to conserve particles and to prevent biasing, the surviving particle weight is multiplied by 10. In theory, it could be argued that this method introduces an additional degree of randomness. In practice, it is a useful mechanism which reduces the number of unimportant calculations thereby saving computing time. This variance reduction technique is usually used in conjunction with other techniques such as splitting.

Splitting Technique: When the splitting technique is included into the model, various importance regions are introduced. As a particle enters a new region, the weight is checked against a value stated for that region. If the particle's weight is greater than the value stipulated, the particle is split into two separate particles, each having half the weight of the original photon. This process is continued until the weights of the new particles are less than the value stated for that region. Since the new particles have half the weight of the original particle, conservation of particle weights is maintained. This method increases the number of particles in the area of interest which in turn decreases the statistical fluctuations.

A particle moving from a less important region to a more important region is split into two or more particles of lower weight. As the particles move to less important regions, Russian Roulette is played to decide if the particle is allowed to survive with augmented weight, or annihilated (12:294).

Transport Function:

The dose rate is related to the photon flux and response function as shown in equation 6a.

$$D = \int_0^{\infty} R(E)\phi(E)dE \quad (6a)$$

where

D = Dose rate

$R(E)$ = Response as a function of energy

$\phi(E)$ = Photon flux as a function of energy

If energy is treated discretely rather than continuously, this can be estimated by the summation shown below.

$$D = \sum_{i=1}^N R_i \phi_i \Delta E_i \quad (6b)$$

This implies that energy is divided in N discrete groups each having energy "width" ΔE_i . The photon flux can be related to the source as,

$$\phi(E) = \int_0^{\infty} t(E' \rightarrow E) q(E') dE' \quad (7a)$$

Where

$q(E')$ = photon source as a function of energy

$t(E' \rightarrow E)$ = Transport function

The transport function $t(E' \rightarrow E)$ converts the source photons with energy E' to the flux of photons arriving at the target with energy E . For each photon source energy, the flux at the target must be evaluated for all other energies.

Again, equation 7a can be written discretely as

$$\phi_i = \sum_{j=1}^N t_{ij} q_j \Delta E_j \quad (7b)$$

The response function is determined by the following equation.

$$R = h\nu \cdot \left(\frac{\mu_a}{\rho} \right) \cdot C \quad (8)$$

where

R = Response function ($\text{rad}/(\gamma\text{-cm}/\text{cm}^3)$)

$h\nu$ = Photon energy (MeV/γ)

$\frac{\mu_a}{\rho}$ = Mass absorption coefficient (cm^2/g) includes scatter loss

C = Conversion factor $1.6021 \times 10^{-18} (\text{rad}/\text{MeV}/\text{g})$

Equation 9a is found by substituting equation 7b into equation 6b. It then can be used to determine the dose rate.

$$D = \sum_{i=1}^N \sum_{j=1}^N (R_i \Delta E_i t_{ij} q_j \Delta E_j) \quad (9a)$$

This can be written in matrix form as follows:

$$D = R \cdot (T \cdot q) \quad (9b)$$

where

R is a 1xN response matrix

T is an NxN transport matrix; $T = t_{ij} \Delta E_i \Delta E_j$,

q is an Nx1 source matrix

If the transport function is known for a given source and target geometry, the dose rate can be determined for any given source spectrum and response function. This allows dose rate calculations to be made for different source spectra without having to re-solve the transport equation (8).

III Method of Analysis

MORSE Model:

The Monte Carlo transport code, MORSE, was used to determine the dose rate one meter above an infinite homogeneous ground plane source. However, before the MORSE code could be used, several subroutines had to be written to model the transport problem (4:31-45, 52-53). These subroutines are included in Appendix A.

A combination of subroutines and input data is used to model a infinite homogeneously distributed ground plane source. To model a homogeneously distributed ground plane source, a boundary crossing flux estimator was used as seen in the BDRYX subroutine. Each crossing of the boundary, set at a height of one meter above the ground plane source, is weighted by a factor of $\left| \frac{1}{\cos \alpha} \right|$ where α is the direction of motion measured away from the vertical. The estimator then determines the flux by dividing the weighted number of particles which cross the boundary by the area of the boundary. In order to model the infinite ground plane source, a one square kilometer plane is used with an albedo surface located on all sides of geometry. The albedo capability refers to the specular, mirror like, reflection of the photons when striking the albedo surface (4:15). Thus, no photons will be lost through the vertical boundaries.

A twenty-one gamma energy group structure was utilized in the MORSE code. This was dictated by the use of the FEWG1 cross-section library which utilizes 21 gamma energy groups (1:4). However, the gamma-ray spectra and gamma emission rates calculated by Millage were accomplished with a nineteen energy group structure (9:117). In the conversion of energy structures from nineteen to twenty-one groups, a major assumption had to be made. It was assumed that each of the nineteen energy bins had an equal gamma emission rate for all energies within the individual group. Thus if the energy bin had to be divided into two equal energy bins in the new structure, half of the gammas would be entered into one bin while the other half would be entered into the other bin. Another assumption was that no fission products emitted a gamma with an energy lower than three kev. This can be substantiated by examining the chart of nuclides (11). The spectra used in this study are seen in Appendix D.

The cross sections and response functions use the twenty-one energy group structure to determine the corresponding dose. Both the cross-sections and response functions which were used in this study come from a standard ANISN format library called FEWG1-85. This state-of-the-art cross section library which contains 37 neutron energy groups and 21 gamma ray energy groups was developed for the Defense Nuclear Agency. FEWG1 cross section library is available through Radiation Shielding

Information Center, RSIC, at Oak Ridge National Laboratory. Again, since this study only pertains to gamma ray transport, the neutron cross sections were not used.

The composition of the air and soil used to determine the appropriate cross sections is listed in Table 1 (10).

Table 1
Composition of Air and Soil Used to Determine Cross-Sections

Element	Air		Soil	
	Atomic Density (Atoms/cm ³)	Fractional Composition	Atomic Density (Atoms/cm ³)	Fractional Composition
H			9.752E+21	.160
N	4.199E+19	.784		
O	1.128E+19	.211	3.480E+22	.570
Al			4.880E+21	.080
Si			1.160E+22	.190
Ar	2.515E+17	.005		

The response function used is the Henderson gamma free-in-air tissue kerma (1:76-79). It has units of rads-tissue/(γ -cm/cm²). Therefore, the dose can be evaluated by

multiplying the response function by the flux. However, the dose calculated by MORSE is normalized per source photon emitted.

Dose Rate:

The first step in calculating the dose rate is to calculate the mono-energetic source conversion coefficient (MSCC). The MSCC is the dose rate (Rad/hr) one meter above a ground plane source with an activity density of one Gamma-Megacurie (3.7×10^{16} gammas per second) distributed homogeneously over one square kilometer. The MSCC is a convenient parameter because it is only a function of the air over ground transport.

The mono-energetic source conversion coefficient was calculated for each of the twenty-one energy groups. This was accomplished by using the MORSE computer code. Since the dose calculated by MORSE (D) is normalized per source photon emitted, equation 10 is used to determine the mono-energetic Source Conversion Coefficient.

$$MSCC = D(\text{rad}/\gamma) \cdot 3600(\text{sec}/\text{hr}) \cdot 3.7 \times 10^{16}(\gamma/\text{sec}) / (MCi/\text{km}^2) \quad (10)$$

After the mono-energetic source conversion coefficient was calculated, the time varying source conversion coefficient (SCC) for a gamma-ray spectrum was determined by multiplying each MSCC by its fraction of occurrence in the source spectrum.

The gamma-ray spectra used in the calculations were determined by Millage. The SCC is a function of time because the source spectrum is a function of time.

$$SCC = \sum_{i=1}^{21} MSCC_i \cdot F_i \quad (11)$$

Once the SCC is calculated, the time dependent Source Normalization "Constant" (K) can be evaluated. The K value is a conversion factor from gamma emission rate to dose rate. It is determined by multiplying the source conversion coefficient by the gamma emission rate. The emission rates used were also those calculated by Millage. The time dependent Source Normalization "Constant" can easily be determined by using the time dependent SCC and GER as seen in equation 12.

$$K(t) = SCC(t) \cdot GER(t) \quad (12)$$

Where

$K(t)$ = Time dependent Source Normalization "Constant"

(rad/hr)/(kt/km²)

$SCC(t)$ = Time dependent Source Conversion Coefficient

(rad/hr)/(MCI/km²)

$GET(t)$ = Time dependent gamma emission rate (MCI/kt)

The final dose rate can be determined by adjusting the K value by other factors such as: fallout fraction, roughness factor, etc.

A unit time Source Normalization Constant can also be found by multiply the SCC evaluated at one hour by the gamma emission rate evaluated at one hour.

$$K_1 = SCC_1 \cdot GER_1 \quad (13)$$

The unit time Source Normalization Constant is often used to approximate the time dependent Source Normalization Constant by the use of the Way-Wigner approximation as seen in equation 14 (6:450-452).

$$K(t) \approx K_1 t^{-1.2} \quad (14)$$

Thus, the unit time K value is the number most often used in ground dose rate calculations.

Transport Matrix:

The Monte Carlo code, MORSE, can also calculate the energy flux for each energy group at the target location. This energy flux is a direct result of the photon source. The photon flux, $\phi(E)$, can be determined by dividing the energy flux by the mid-point energy of each energy group. Each energy group was used as the source to establish the photon flux distribution at the target location. These results were then used to evaluate a

transport matrix. See Appendix C for an example of the calculations made to determine the transport matrix. This matrix can be used for any given photon source spectrum and response function to calculate the total response due to a ground plane source.

IV Results and Discussion

The Source Conversion Coefficients for mono-energetic photons were determined by using the Monte Carlo transport code, Morse. The SCC is based on a source of one gamma-Megacurie homogeneously spread over one square kilometer. A gamma-megacurie is defined to be 3.7×10^{16} gamma rays per second. Figures 4 and 5 show the source conversion coefficients for mono-energetic photon sources. The values are plotted at the midpoint of each energy group. All values have a statistical error less than one percent. These values, also seen in Appendix F, can be used to determine the SCC for a gamma-ray spectrum.

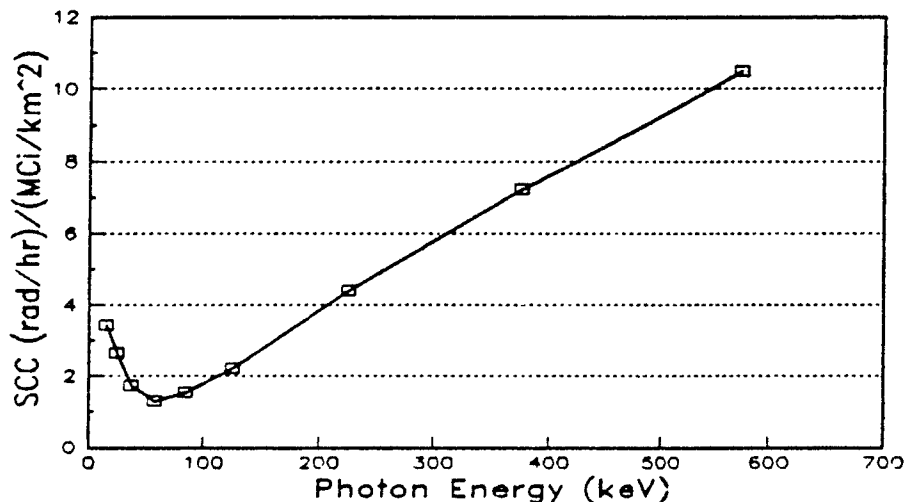


Figure 4
Mono-energy Gamma-ray Source Conversion Coefficients
for low Gamma-ray Energies

As seen in Figure 4, the SCC for the lower energy photons actually increases due to the increase in the response functions at the lower energies. This increase is due to the rapidly increasing absorption cross sections due to the photoelectric effect at those energies.

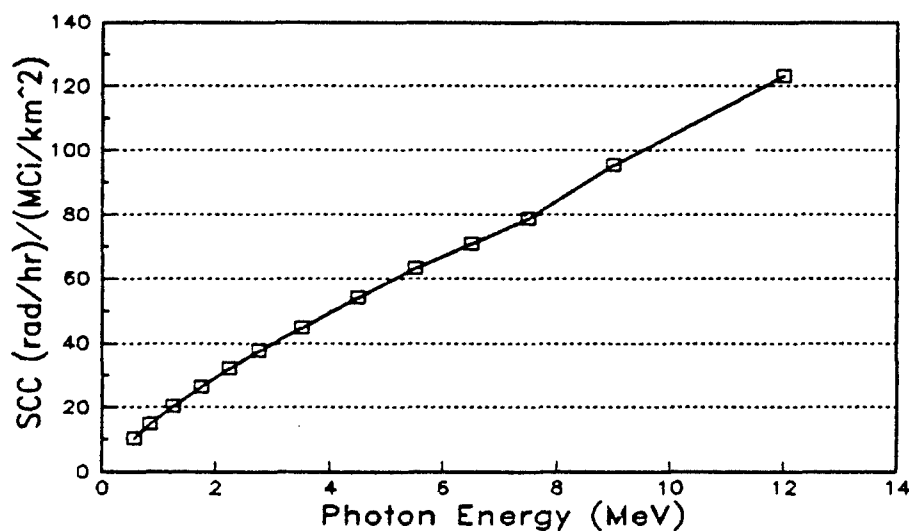


Figure 5
Mono-energy Gamma-ray Source Conversion Coefficients
for high Gamma-ray Energies

Two types of Monte Carlo calculations were made to compare how different assumptions affect the SCC. One calculation assumed there was a homogeneous plane source in an infinite medium of air. This is the same assumption made in the BUF method. Figure 6 shows how Millage's evaluation of the SCC

compares to the Monte Carlo method. The difference between the BUF method and Monte Carlo with infinite air is approximately 3 percent.

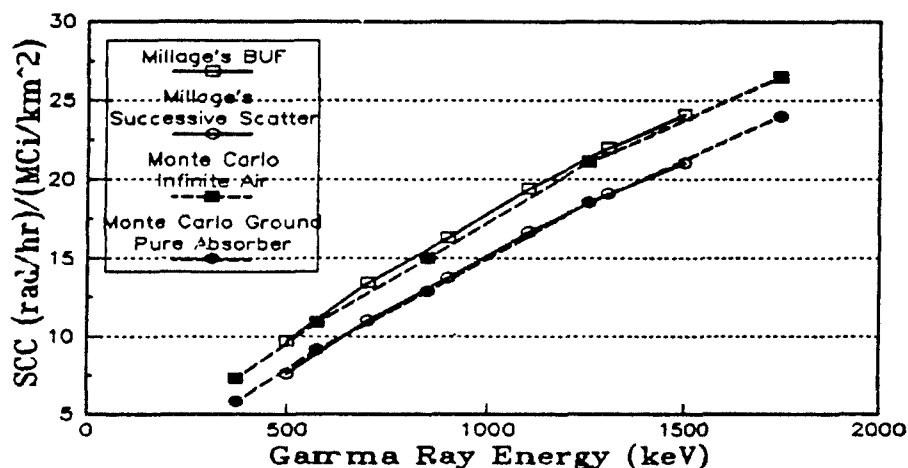


Figure 6
Comparison of Mono-energy Gamma-ray Source Conversion Coefficients as calculated by different methods

Another Monte Carlo calculation was made for a homogeneous ground plane source where the ground was modeled as a pure absorber. This assumption was also used in the Successive Scatter Technique performed by Millage. Figure 6 also shows how this calculation compares to Millage's evaluation of the SCC. There is less than a 1 percent difference between the Successive Scatter method and the assumption of the ground being a pure absorber in the Monte Carlo method. Thus, no

difference can be determined since the Monte Carlo values have a statistical error approximately equal to 1 percent. This verifies the Successive Scatter methodology is an accurate procedure in determining the contribution of scattered photons to dose rate. However, the assumption used in determination of the Source Conversion Coefficient is the major factor which dictates the end result of the SCC. Both methods used by Millage, BUF and Successive Scatter, are an accurate representation of the true dose rates given the assumptions used.

Effects of the assumptions made by Millage in his SCC calculations can be seen in Figure 7. As seen in the plot, the assumption of infinite air does not affect the true dose rate to a high degree. The maximum error in the infinite air dose rate is approximately 5 percent. Whereas, the error associated with the assumption that the ground is a pure absorber is approximately 14 percent.

The time dependent SCC for the fuels of U-235, U-238, and Pu-239 can be seen in Figure 8. The variations are caused by the distinct energy spectra and gamma emission rates for the fuels as a function of time.

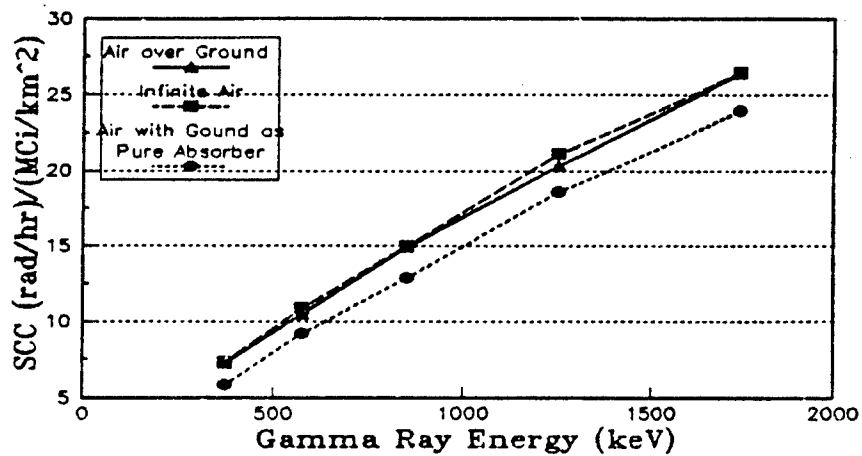


Figure 7
Comparison of Mono-energy Gamma-ray Source Conversion Coefficients calculated by Monte Carlo

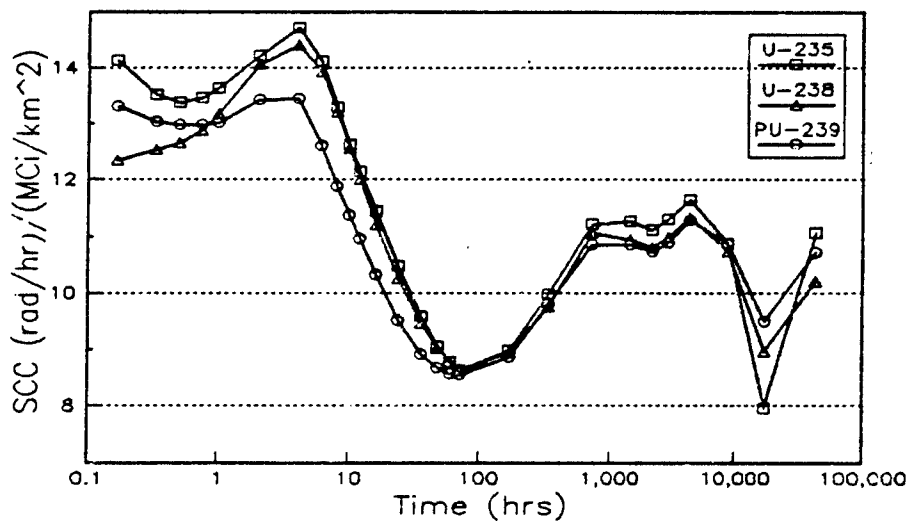


Figure 8
Time dependent Source Conversion Coefficient for U-235, U-238, Pu-239

The source normalization constant (K) was determined by multiplying the SCC and GER together. Table 2 shows the K_1 value one hour after detonation for the three fuel types. It compares these values to those which were presented by Glasstone and Millage.

Table 2
Source Normalization Constant 1 hour after detonation

	(rad/hr)/(kt/km ²)		
Fuel	Monte Carlo	Millage	Glasstone
U-235	8064	7059	7186
U-238	8467	7369	7186
Pu-239	7666	6684	7186

As expected, the K_1 values determined by the Monte Carlo calculations are higher than the values which Millage obtained. Since both methods use the same gamma emission rate, the difference is due to the assumption which was made to calculate the Source Conversion Coefficients. Once more, the values obtained by Millage were determined from the Successive Scatter technique. This method uses the assumption that the ground is a pure absorber which has already been shown to be in error.

Figures 9, 10, and 11 show how the Source Normalization "Constant" varies as a function of time. The ratio of calculated K values are compared with the Way-Wigner approximation

in Figure 13. As seen in Figure 12, the Way-Wigner approximation overestimates the dose rate by as much as twice that calculated by the MORSE code with an average K value approximately 0.75 of the Way-Wigner estimation. Thus, the Way-Wigner approximation should only be used for rough estimations of the dose rate. After six months, 43200 hours, the difference increases rapidly. However, this is expected since it was stated earlier that the Way-Wigner approximation is only valid up to six months.

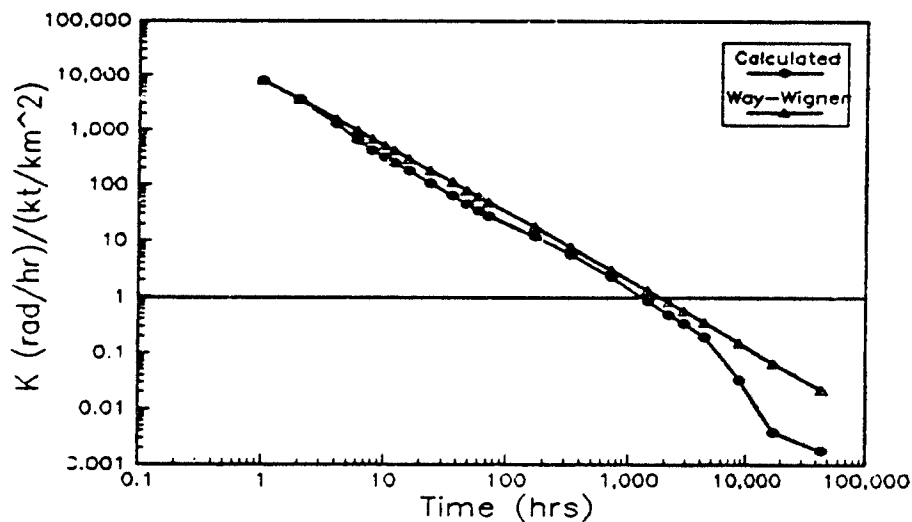


Figure 9
Time dependent Source Normalization Constant
vs Way-Wigner Approximation for U-235

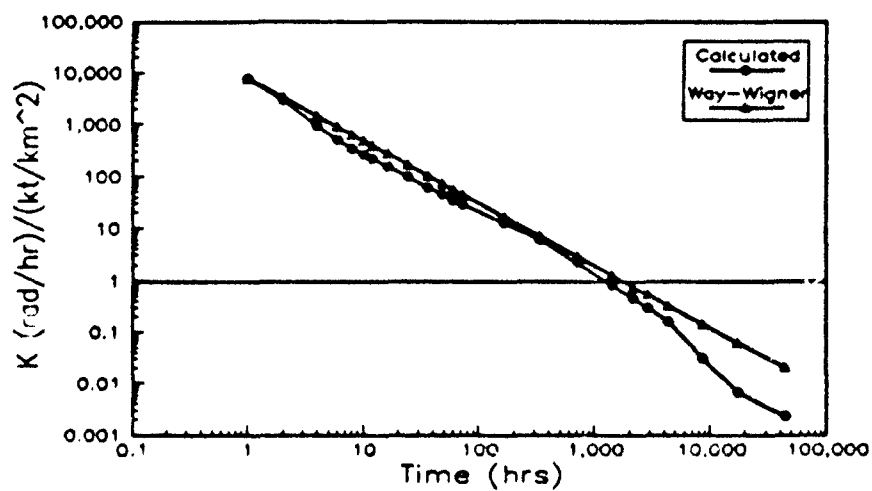


Figure 10
Time dependent Source Normalization Constant
vs Way-Wigner Approximation for U-238

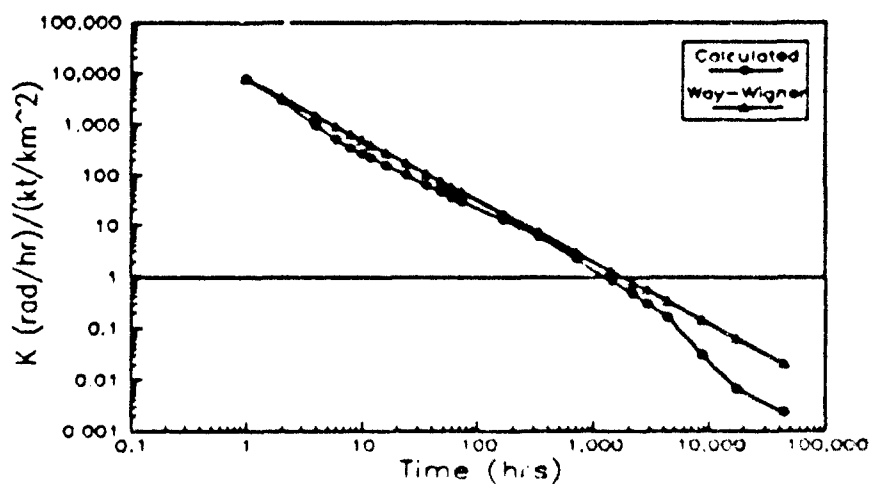


Figure 11
Time dependent Source Normalization Constant
vs Way-Wigner Approximation for Pu-239

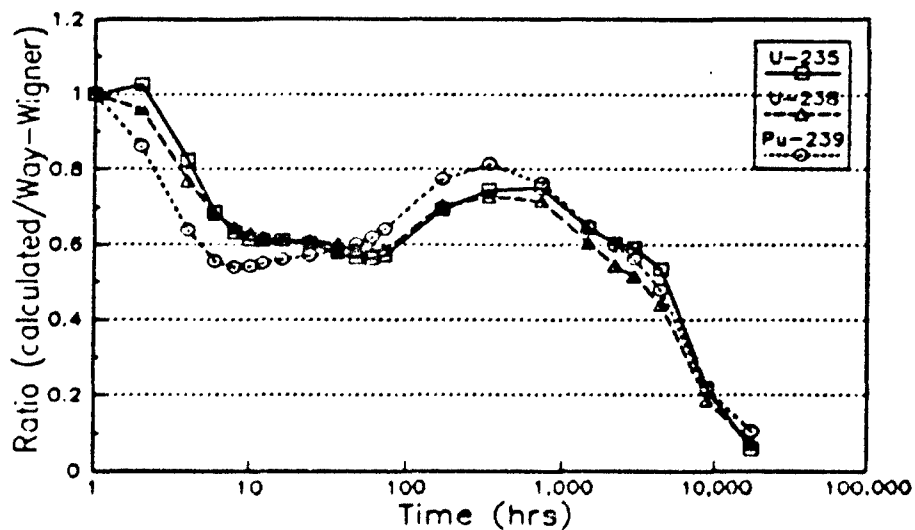


Figure 12
Ratio of Calculated Source Normalization Constant
to the Way-Wigner Approximation

The transport matrix was determined by using MORSE to calculate the associated energy flux distribution caused by a source photon from each of the 21 energy bins. This generated a 21 x 21 transport matrix, as seen in Appendix E. The transport matrix can be used with any source spectrum and response function to obtain a dose rate without re-solving the transport problem or making more computer runs.

V Conclusions and Recommendations:

The purposes of this study were as follows.

- (1) Evaluation of the dose rate from a homogeneous ground plane source as determined by the fission products of three major fuels, U-235, U-238, and Pu-239.
- (2) Examination of the major assumptions made by Millage and their effect on the dose rate.
- (3) Evaluation of a transport matrix which will convert any given gamma-ray spectrum and response function to a ground dose rate.

The dose rates were calculated by a Monte Carlo method. MORSE, a Monte Carlo neutron and gamma-ray transport code developed by Oak Ridge National Laboratories, was utilized for this purpose. The FEWG1 library, a state-of-the-art cross section library developed for the Defense Nuclear Agency consisting of 21 gamma-ray energy groups, was used to calculate the cross sections and response functions utilized by the MORSE code.

In order to evaluate the dose rate, the mono-energetic Source Conversion Coefficients were calculated from MORSE output. The Source Conversion Coefficient is determined by using an infinite plane source with an activity density of one Gamma-Megacurie homogeneously distributed over one square kilometer. An appropriate number of runs were used in order to

obtain a statistical fluctuation less than one percent. The mono-energetic SCC's were then used in conjunction with the gamma-ray spectra to obtain a time dependent Source Conversion Coefficient for each of the three major fuels. However, the mono-energetic SCC in Appendix F can be used to determine the SCC for any given spectrum. The gamma-ray spectra which were utilized in the calculations were acquired from Millage's data.

The Source Normalization Constant was also calculated by the use of the gamma-emission rates determined by Millage. Values obtained for the Source Normalization Constant one hour after burst are 8064 (rad/hr)/(kt/km²) for U-235, 8467 (rad/hr)/(kt/km²) for U-238, and 7666 (rad/hr)/(kt/km²) for Pu-239. The Source Normalization values can then be adjusted for fallout fraction, terrain roughness, etc, to obtain the true dose rate. The time dependent K value should be used if possible due to the difference, up to 95 percent, between the calculated value and the Way-Wigner approximation. If the unit time Source Normalization Constant is used, a correction factor as seen in Figure 13 should be used to obtain better results.

Further studies of the gamma-ray spectra should be accomplished. This includes the spectra which result from induced activity from the weapon and the distribution of photons within each of the energy groups used by Millage. This could verify or disprove the assumption that the spectrum was uniformly distributed within an energy group. The assumption was used when

the 19 group structure was converted to the 21 group structure. Another approach to this problem could be the use of MCNP, a continuous energy Monte Carlo code. However if a different spectrum is determined for the source, the transport matrix found in this study can be used to find the appropriate dose rate.

Appendix A

Morse Source Code for User Written Routines

A.1 MORSEB.FOR

```
      Program MORSEB.FOR
C * * THIS IS THE MAIN ROUTINE * * * * *
C * * Modified by Mike Howard Oct 1989 for infinite ground plane source * *
C * * dose calculations
C * * THE FOLLOWING CARD DETERMINES THE SIZE ALLOWED FOR BLANK COMMON * * *
      COMMON NC(200000)
C * * (REGION SIZE NEEDED IS ABOUT 150K + 4*(SIZE OF BLANK COMMON IN WORDS))
C * * NOTE - THE ORDER OF COMMONS IN THIS ROUTINE IS IMPORTANT AND MUST
C           CORRESPOND TO THE ORDER USED IN DUMP ROUTINES SUCH AS HELP,
C           XSCHLP, AND USRHLP
C * *
C * * LABELLED COMMONS FOR WALK ROUTINES * * * * *
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(26),ITOUT,ITIN
      COMMON /FISBNK/ MFISTP
      COMMON /NUTRON/ NAME
C * *
C * * LABELLED COMMONS FOR CROSS-SECTION ROUTINES * * * * *
      COMMON /LOCSIG/ ISCCOG
      COMMON /MEANS/ NM
      COMMON /MOMENT/ NMOM
      COMMON /QAL/ Q
      COMMON /RESULT/ POINT
C * *
C * * LABELLED COMMONS FOR GEOMETRY INTERFACE ROUTINES * * * * *
      COMMON /GEOMC/ XTWO
      COMMON /NORMAL/ UNORM
C * *
C * * LABELLED COMMONS FOR USER ROUTINES * * * * *
      COMMON /PDET/ ND
      COMMON /USER/ AGST
C * *
C * * COMMON /DUMMY/ WILL NOT BE FOUND ELSEWHERE IN THE PROGRAM * * * * *
      COMMON /DUMMY/ DUM
C * *
      CHARACTER*20, NAM1
      CHARACTER*20, NAM2
      TYPE *, 'YOU ARE USING MORSEB WITH A POINT SOURCES'
      TYPE *, 'THIS VERSION USES A POINT SOURCE AT THE ORIGIN.'
      TYPE *, 'ENTER NAME OF INPUT FILE:'
      ACCEPT 100, NAM1
100  FORMAT(A20)
      TYPE 101, NAM1
```

```

101  FORMAT(X,A20)
      TYPE *, 'ENTER NAME OF OUTPUT FILE:'
      ACCEPT 200, NAM2
      TYPE 101, NAM2
200  FORMAT (A20)
      OPEN(UNIT=1,NAME=NAM1,TYPE='OLD')
      OPEN(UNIT=2,NAME=NAM2,TYPE='NEW')
      ITOUT = 2
      ITIN = 1
      NLFT=199999
      CALL MORSE(NLFT)
      TYPE 300, NAM2
300  FORMAT(X,'OUTPUT FILE IS ',A20)
      STOP
      END

```

A.2 BANKR

```

SUBROUTINE BANKR(NBNKID)
C DO NOT CALL EUCLID FROM BANKR(7)
COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
2 IO,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGFQTG,NGPQTN,NITS,
5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMOST,NMTG,NOLEAK,NORMF,NPAST,
6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
COMMON /MUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
NBNK = NBNKID
IF (NBNK) 100,100,140
100 NBNK = NBNK + 5
GO TO (104,103,102,101),NBNK
101 CALL STRUN
RETURN
102 NBAT = NITS - ITERS
NSAVE = NMEM
CALL STBTCH(NBAT)
C NBAT IS THE BATCH NO. LESS ONE
RETURN
103 CALL NBATCH(NSAVE)
C NSAVE IS THE NO. OF PARTICLES STARTED IN THE LAST BATCH
RETURN
104 CALL NRUN(NITS,NQUIT)
C NITS IS THE NO. OF BATCHES COMPLETED IN THE RUN JUST COMPLETED
C NQUIT .GT. 1 IF MORE RUNS REMAIN
C .EQ. 1 IF THE LAST SCHEDULED RUN HAS BEEN COMPLETED
C IS THE NEGATIVE OF THE NO. OF COMPLETE RUNS, WHEN AN
EXECUTION TIME KILL OCCURS
RETURN
140 GO TO (1,2,3,4,5,6,7,8,9,10,11,12,13),NBNK
C NBNKID COLL TYPE BANKR CALL NBNKID COLL TYPE BANKR CALL
C 1 SOURCE YES (MSOUR) 2 SPLIT NO (TESTW)
C 3 FISSION YES (FPROB) 4 GAMGEN YES (GSTORE)
C 5 REAL COLL YES (MORSE) 6 ALBEDO YES (MORSE)
C 7 BDRYX YES (NXTCOL) 8 ESCAPE YES (NXTCOL)
C 9 E-CUT NO (MORSE) 10 TIME KILL NO (MORSE)
C 11 R R KILL NO (TESTW) 12 R R SURV NO (TESTW)
C 13 GAMLOST NO (GSTORE)
1 RETURN
2 RETURN
3 RETURN
4 RETURN
5 RETURN
6 RETURN
7 CALL BDRYX

```

RETURN
8 RETURN
9 RETURN
10 RETURN
11 RETURN
12 RETURN
13 RETURN
END

A.3 GETMED

```
SUBROUTINE GTMED(MDGEOM,MDXSEC)
IF (MDGEOM.LT.2) THEN
  MDXSEC=1
ELSE
  MDXSEC = 2
ENDIF
RETURN
END
```

A.4 BDRYX

```
SUBROUTINE BDRYX
C IDENTIFIES DETECTOR POSITION WITH A BOUNDARY CROSSING AND THEN
C CALCULATES AND SUMS QUANTITIES OF INTEREST FOR EACH BATCH.
C
COMMON /USER/ AGSTRT,WTSTRT,XSTRT,YSTRT,ZSTRT,DFP,EBOTN,EBOTG,
1 TCUT,I0,I1,IADJM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,NLAST,
2 NLEFT,NMGP,NMTG,NSTRT
COMMON /PDET/ ND,NNE,NE,NT,NA,NRESP,NEX,NEXND,NEND,NDNR,NTNR,NTNE,
1 NAME,NTNDNR,NTNEND,NANEND,LOCRSP,LOCXD,LOCIB,LOCCO,LOCT,LOCUD,
2 LOCSD,LOCQE,LOCQT,LOCQTE,LOCQAE,LMAX,EFIRST,EGTOP
COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLDBDRY
1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGEBDRY
DATA ABDX/1.0E10/,ZBDX/100./
Z1 = ZBDX+.1
Z2 = ZBDX-.1
IF ((Z.GT.Z2).AND.(Z.LT.Z1)) THEN
  ABCOS=ABS (W)
IF (ABCOS.LT.0.01) ABCOS=0.005
CON=WATE/ABCOS/ABDX
CALL FLUXST(1,IG,CON,AGE,0,0)
ENDIF
RETURN
END
```

A.5 SOURCE

```

SUBROUTINE SOURCE(IG,U,V,W,X,Y,Z,WATE,MED,AG,ISOUR,ITSTR,NGPQT3,
1DDF,ISBIAS,NMTG)
COMMON /USER/ DUM(9),IO,I1,IDUM(12)
COMMON WTS(1)
C
C IF ITSTR=0, MUST PROVIDE IG,X,Y,Z,U,V,W,WATE AND AG IF DESIRED TO BE
C DIFFERENT FROM CARD VALUES (WHICH ARE THE VALUES INPUT TO SOURCE)
C IF ITSTR=1, IG IS THE GRP NO. CAUSING FISSION, MUST PROVIDE NEW IG
C THIS VERSION OF SOURCE SELECTS INITIAL GROUP FROM THE INPUT SPEC
C
DATA ICALL/1/
IF (ICALL) 10,10,5
5 ICALL = 0
WRITE (IO,1000)
1000 FORMAT (' YOU ARE USING THE DEFAULT VERSION OF SOURCE WHICH SETS
1WATE TO DDF AND PROVIDES AN ENERGY IG.')
10 IF(ISOUR)15,15,60
15 WATE=DDF
IF (ISBIAS) 20,20,25
20 NWT = 2*NMTG
GO TO 30
25 NWT = 3*NMTG
30 R = FLTRNF(0)
DO 35 I=1,NGPQT3
IF (R - WTS(I+NWT)) 40,40,35
35 CONTINUE
40 IG=I
IF (ISBIAS) 60,60,45
45 IF (I-1) 60,50,55
50 WATE = WATE*WTS(2*NMTG+1)/WTS(3*NMTG+1)
GO TO 60
55 WATE = WATE*(WTS(2*NMTG+I)-WTS(2*NMTG+I-1))/(WTS(3*NMTG+I)-WTS(3*N
1MTG+I-1))
C *****
C This version of the MORSE SOURCE file is for a POINT source
c AT THE ORIGIN.
C *****
60 Z = 0.0
Y = 0.0
X = 0.0
100 RETURN
END

```

Appendix B

The input file used to determine the MSCC for a gamma source with energy from group 14. The important results from the output is also listed in this appendix. All other runs were made with a different source energy.

B.1 MORSE Input File

```

DISTRIBUTED SOURCE, AIR & GROUND, DOSE AT 1m ABOVE GROUND, MORSE
1000 4001 100 1 0 21 21 21 0 0 450. 2 3
14 1 0 0 1.0 0 1.0E+4 0.0 0
0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0
1.4000 +7 1.0000 +7 8.0000 +6 7.0000 +6 6.0000 +6 5.0000 +6 4.0000 +6
3.0000 +6 2.5000 +6 2.0000 +6 1.5000 +6 1.0000 +6 7.0000 +5 4.5000 +5
3.0000 +5 1.5000 +5 1.0000 +5 7.0000 +4 4.5000 +4 3.0000 +4 2.0000 +4
0000DF4FC1C2
1 1 0 0 0 7 21
0 0 0 1 1 1 0.5 0.01 1.0 0.0
0 0 0 2 1 2 .75 0.05 1.0 0.0
0 0 0 3 1 3 1.0 0.05 1.0 0.0
0 0 0 4 1 4 1.0 0.10 1.0 0.0
0 0 0 5 1 5 1.0 0.10 1.0 0.0
0 0 0 6 1 6 1.0 0.10 1.0 0.0
-1 0 0 0 0
0 0
TEST CASE FOR THESIS
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 1.0E+2
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 0.1E+5
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 0.5E+5
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 1.0E+5
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 3.0E+5
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.0E-9 4.0E+5
RPP -5.0E+4 5.0E+4 -5.0E+4 5.0E+4 -1.5E+3 4.0E+5
RPP -5.5E+4 5.5E+4 -5.5E+4 5.5E+4 -1.5E+3 4.0E+5
RPP -6.5E+7 6.5E+7 -6.5E+7 6.5E+7 -5.5E+7 5.5E+7
END
1 +1
2 +2 -1
3 +3 -2
4 +4 -3
5 +5 -4
6 +6 -5
7 +7 -6
8 +8 -7
9 +9 -8
END
1 1 2 3 4 5 6 7 7
1 1 1 1 1 1 2 3 0

```

```

21 GROUP GAMMA CROSS SECTIONS ---- AIR AND GROUND---ENC 3
  0  0  21  21  58  61  4  2  6  7  4  1  1
  1  0  0  0  0  0  0 -10  0  0  0
SAMBO ANALYSIS INPUT DATA FOR SCC (GAMMAS)
  1  0  21  0  0  1  1  1
    0.000      +0.0 1.0000E+2
      (NORMALIZED PER SINGLE GAMMA FROM THE SOURCE)
PHOTON DOSE (RADS TISSUE/GAMMA/CM^2)
3.2081E-9 2.4722E-9 2.0847E-9 1.8651E-9 1.6613E-9 1.4431E-9 1.1971E-9
1.0111E-9 8.707E-10 7.253E-10 5.641E-10 4.106E-10 2.930E-10 1.923E-10
1.106E-10 5.482E-11 3.711E-11 3.472E-11 6.327E-11 1.416E-10 4.406E-10
(gammas/cm^2 ev)
  1  2  3  4  5  6  7  8  9  10  11  12  13  14
 15 16 17 18 19 20 21
$$$$$$$$$ MORSE6.FOR PROBLEM ***** MONO.DAT *****

```

B.2 Results from MORSE Output

PHOTON DOSE (RADS TISSUE/GAMMA/CM²)
 RESPONSES (DETECTOR) (NORMALIZED PER SINGLE GAMMA FROM THE SOURCE)

DETECTOR	UNCOLL	FSD	TOTAL	FSD
1	0.0000E+00	0.00000	5.4267E-20	0.00706

DETECTOR NO.	1	FLUENCE (ENERGY, DETECTOR) (gammas/cm ² ev)
ENERGIES		
1.000E+06		0.000E+00 0.000
7.000E+05		0.000E+00 0.000
4.500E+05		1.425E-15 0.009
3.000E+05		5.032E-16 0.008
1.500E+05		8.384E-16 0.010
1.000E+05		1.135E-15 0.012
7.000E+04		1.019E-15 0.010
4.500E+04		3.472E-16 0.021
3.000E+04		2.697E-17 0.043
2.000E+04		1.842E-19 0.343
1.000E+04		

APPENDIX C

A sample calculation of the Mono-Energetic Source Conversion Coefficient is presented along with a sample of how the Transport Matrix was determined.

Mono-Energetic Source Conversion Coefficient:

From output in Appendix B.

Dose = $5.4267\text{E-}20$ rad-tissue/per gamma emitted

MSCC = Dose * 3.7×10^{10} (gammas/sec)/(MCI/km²) * 3600 sec/hr

MSCC = 7.2 (rad-tissue/hr)/(MCI/km²)

In order to determine the Transport Matrix, must use the output from Appendix B.

$T_{1,1} = f_1$ (ev/cm²) * $\Delta E_1 \cdot \Delta E_1$,

where $f_{1,1} = 5.032\text{E-}16$ ev/cm²

ΔE_1 = difference between upper and lower bound of the bin being examined. (3×10^5 ev - 1.5×10^5 ev = 1.5×10^5 ev)

ΔE_1 = difference between upper and lower bound of the energy bin of source particle. (4.5×10^5 ev - 3.0×10^5 ev = 1.5×10^5 ev)

Using these numbers:

$T_{1,1,1} = 1.13 \times 10^{-9}$ gammas/cm²

This value can be verified by looking at the Transport Matrix column 14 row 15 in Appendix E. This procedure is repeated

for every energy bin of the source particle and for each bin within the energy of the source particle until the 21 x 21 matrix is complete.

APPENDIX D

The following tables list the Gamma Emission Rate (GER) per second per fission event of the fuel for each energy bin. The GER was used to determine the fraction from each bin used in the time dependent Source Conversion Coefficient. The Source Normalization Constant was also calculated with the use of the GER. There are three fuels presented in this section, U-235, U-238, and Pu-239.

D.1 U-235

Energy (MeV)	10 min	20 min	30 min	45 min	1 hour
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	3.251E-09	1.795E-11	1.162E-13	1.726E-16	1.354E-18
6.500	6.502E-09	3.590E-11	2.324E-13	3.452E-16	2.708E-18
5.500	3.498E-07	1.526E-08	9.142E-10	2.685E-11	1.721E-12
4.500	3.649E-06	2.764E-07	4.549E-08	1.900E-08	1.407E-08
3.500	1.186E-05	5.423E-06	2.846E-06	1.331E-06	8.040E-07
2.750	1.869E-05	7.677E-06	4.794E-06	3.222E-06	2.403E-06
2.250	3.897E-05	1.985E-05	1.334E-05	8.959E-06	6.523E-06
1.750	6.387E-05	3.352E-05	2.157E-05	1.380E-05	1.004E-05
1.250	1.332E-04	7.384E-05	4.779E-05	2.965E-05	2.142E-05
0.850	2.051E-04	1.093E-04	7.204E-05	4.843E-05	3.766E-05
0.575	1.402E-04	7.425E-05	4.880E-05	3.185E-05	2.384E-05
0.375	1.061E-04	6.340E-05	4.199E-05	2.564E-05	1.746E-05
0.225	1.161E-04	6.971E-05	4.673E-05	2.856E-05	1.915E-05
0.125	3.800E-05	2.229E-05	1.518E-05	9.647E-06	6.680E-06
0.085	2.160E-05	1.161E-05	7.359E-06	4.450E-06	3.099E-06
0.058	1.800E-05	9.679E-06	6.133E-06	3.709E-06	2.583E-06
0.038	1.080E-05	5.807E-06	3.680E-06	2.225E-06	1.550E-06
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	2 hours	4 hours	6 hours	8 hours	10 hours
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	3.347E-20	1.425E-21	3.412E-22	1.742E-22	1.369E-22
6.500	6.693E-20	2.849E-21	6.824E-22	3.484E-22	2.738E-22
5.500	6.302E-16	2.655E-17	5.196E-18	1.214E-18	2.885E-19
4.500	5.987E-09	2.409E-09	1.364E-09	8.053E-10	4.782E-10
3.500	3.280E-07	9.866E-08	3.409E-08	1.387E-08	6.560E-09
2.750	9.486E-07	2.807E-07	1.269E-07	6.784E-08	3.910E-08
2.250	2.501E-06	9.184E-07	4.826E-07	2.710E-07	1.576E-07
1.750	4.134E-06	1.573E-06	8.302E-07	5.028E-07	3.401E-07
1.250	1.020E-05	3.990E-06	2.110E-06	1.367E-06	9.870E-07
0.850	1.919E-05	6.310E-06	2.772E-06	1.685E-06	1.274E-06
0.575	1.115E-05	4.049E-06	2.441E-06	1.910E-06	1.631E-06
0.375	5.923E-06	1.554E-06	9.416E-07	7.719E-07	6.729E-07
0.225	6.080E-06	1.650E-06	9.852E-07	7.532E-07	6.246E-07
0.125	2.377E-06	8.477E-07	4.963E-07	3.330E-07	2.453E-07
0.085	1.145E-06	3.816E-07	2.627E-07	2.156E-07	1.837E-07
0.058	9.543E-07	3.180E-07	2.189E-07	1.796E-07	1.531E-07
0.038	5.726E-07	1.908E-07	1.314E-07	1.078E-07	9.184E-08
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	12 hours	16 hours	24 hours	36 hours	48 hours
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	1.276E-22	1.222E-22	1.129E-22	9.669E-23	8.123E-23
6.500	2.552E-22	2.444E-22	2.258E-22	1.934E-22	1.625E-22
5.500	7.095E-20	7.651E-21	3.625E-21	3.113E-21	2.651E-21
4.500	2.847E-10	1.016E-10	1.328E-11	6.740E-13	3.696E-14
3.500	3.440E-09	1.093E-09	1.294E-10	9.620E-12	5.331E-12
2.750	2.349E-08	9.205E-09	1.963E-09	6.707E-10	5.792E-10
2.250	9.476E-08	3.781E-08	8.984E-09	2.909E-09	2.112E-09
1.750	2.497E-07	1.549E-07	7.325E-08	2.945E-08	1.722E-08
1.250	7.509E-07	4.676E-07	2.153E-07	9.435E-08	5.646E-08
0.850	1.065E-06	8.234E-07	5.486E-07	3.429E-07	2.418E-07
0.575	1.435E-06	1.151E-06	7.991E-07	5.140E-07	3.560E-07
0.375	5.967E-07	4.817E-07	3.376E-07	2.217E-07	1.577E-07
0.225	5.411E-07	4.366E-07	3.202E-07	2.237E-07	1.684E-07
0.125	1.957E-07	1.472E-07	1.096E-07	8.183E-08	6.497E-08
0.085	1.599E-07	1.279E-07	9.857E-08	8.545E-08	8.030E-08
0.058	1.332E-07	1.066E-07	8.214E-08	7.121E-08	6.691E-08
0.038	7.993E-08	6.394E-08	4.929E-08	4.272E-08	4.015E-08
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	60 hours	72 hours	7 days	14 days	30 days
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	6.775E-23	5.640E-23	1.289E-23	9.753E-25	2.673E-27
6.500	1.355E-22	1.128E-22	2.578E-23	1.951E-24	5.345E-27
5.500	2.238E-21	1.878E-21	4.284E-22	6.705E-23	1.362E-23
4.500	2.156E-15	1.314E-16	3.938E-21	3.218E-22	1.051E-24
3.500	5.574E-12	5.903E-12	6.537E-12	4.986E-12	2.105E-12
2.750	5.746E-10	5.750E-10	5.405E-10	4.119E-10	1.794E-10
2.250	1.912E-09	1.800E-09	1.218E-09	7.033E-10	2.679E-10
1.750	1.479E-08	1.508E-08	1.596E-08	1.065E-08	4.427E-09
1.250	4.166E-08	3.487E-08	2.129E-08	1.181E-08	4.518E-09
0.850	1.854E-07	1.504E-07	5.704E-08	2.111E-08	2.075E-09
0.575	2.609E-07	2.008E-07	6.851E-08	3.102E-08	1.419E-08
0.375	1.188E-07	9.364E-08	3.449E-08	1.612E-08	5.417E-09
0.225	1.338E-07	1.106E-07	4.258E-08	1.525E-08	4.776E-09
0.125	5.373E-08	4.577E-08	1.813E-08	5.680E-09	2.254E-09
0.085	7.575E-08	7.084E-08	3.615E-08	1.377E-08	3.008E-09
0.058	6.313E-08	5.904E-08	3.013E-08	1.147E-08	2.506E-09
0.038	3.788E-08	3.542E-08	1.808E-08	6.883E-09	1.504E-09
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	60 days	90 days	120 days	180 days	1 year
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	4.195E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00
6.500	8.389E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00
5.500	1.019E-24	7.631E-26	5.713E-27	3.202E-29	0.000E+00
4.500	2.293E-29	5.004E-34	0.000E+00	0.000E+00	0.000E+00
3.500	4.069E-13	8.060E-14	1.722E-14	2.025E-15	9.369E-16
2.750	3.457E-11	6.716E-12	1.322E-12	6.453E-14	1.020E-14
2.250	5.878E-11	1.875E-11	1.048E-11	7.530E-12	4.752E-12
1.750	9.164E-10	1.810E-10	3.489E-11	2.587E-12	9.890E-13
1.250	9.352E-10	1.919E-10	4.231E-11	6.580E-12	2.121E-12
0.850	5.218E-09	3.846E-09	2.969E-09	1.752E-09	3.045E-10
0.575	7.098E-09	4.790E-09	3.505E-09	1.939E-09	3.182E-10
0.375	1.183E-09	4.178E-10	2.010E-10	6.475E-11	4.854E-12
0.225	1.469E-09	6.483E-10	3.325E-10	1.180E-10	3.420E-11
0.125	1.126E-09	5.877E-10	3.193E-10	1.161E-10	3.397E-11
0.085	5.784E-10	2.533E-10	1.385E-10	6.642E-11	3.328E-11
0.058	4.820E-10	2.111E-10	1.154E-10	5.535E-11	2.774E-11
0.038	2.892E-10	1.266E-10	6.926E-11	3.321E-11	1.664E-11
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	2 years	5 years
12.000	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00
7.500	0.000E+00	0.000E+00
6.500	0.000E+00	0.000E+00
5.500	0.000E+00	0.000E+00
4.500	0.000E+00	0.000E+00
3.500	4.720E-16	6.039E-17
2.750	5.022E-15	6.167E-16
2.250	1.953E-12	1.356E-13
1.750	4.032E-13	3.060E-14
1.250	8.287E-13	8.377E-14
0.850	2.650E-11	1.675E-11
0.575	3.037E-11	1.737E-11
0.375	1.348E-12	2.230E-13
0.225	1.311E-11	9.493E-13
0.125	1.309E-11	9.394E-13
0.085	1.361E-11	1.972E-12
0.058	1.134E-11	1.644E-12
0.038	6.804E-12	9.861E-13
0.025	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00

D.2 Pu-239

Energy (MeV)	10 min	20 min	30 min	45 min	1 hour
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	1.821E-09	1.255E-11	9.956E-14	7.472E-17	1.501E-13
6.500	3.642E-09	2.511E-11	1.991E-13	1.494E-16	3.001E-18
5.500	1.105E-07	3.722E-09	2.909E-10	1.626E-11	1.094E-12
4.500	1.606E-06	9.413E-08	2.124E-08	1.183E-08	8.153E-09
3.500	1.119E-05	4.946E-06	3.040E-06	1.698E-06	1.012E-06
2.750	1.563E-05	8.355E-06	5.254E-06	3.014E-06	2.004E-06
2.250	3.784E-05	2.104E-05	1.385E-05	8.073E-06	5.143E-06
1.750	6.540E-05	3.594E-05	2.385E-05	1.446E-05	9.603E-06
1.250	1.274E-04	6.708E-05	4.492E-05	2.834E-05	1.957E-05
0.850	1.768E-04	1.011E-04	7.029E-05	4.808E-05	3.642E-05
0.575	1.384E-04	7.468E-05	5.100E-05	3.358E-05	2.422E-05
0.375	1.185E-04	7.506E-05	5.198E-05	3.172E-05	2.047E-05
0.225	1.318E-04	8.170E-05	5.575E-05	3.382E-05	2.192E-05
0.125	4.247E-05	2.288E-05	1.486E-05	9.237E-06	6.457E-06
0.085	2.946E-05	1.360E-05	8.282E-06	4.809E-06	3.264E-06
0.058	2.455E-05	1.134E-05	6.901E-06	4.008E-06	2.720E-06
0.038	1.473E-05	6.801E-06	4.141E-06	2.405E-06	1.632E-06
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	2 hours	4 hours	6 hours	8 hours	10 hours
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	1.130E-19	3.851E-21	9.136E-22	6.894E-22	6.089E-22
6.500	2.261E-19	7.703E-21	1.827E-21	1.379E-21	1.218E-21
5.500	2.085E-15	4.702E-17	7.098E-18	1.633E-18	3.927E-19
4.500	3.242E-09	1.320E-09	6.404E-10	3.349E-10	1.853E-10
3.500	2.463E-07	7.277E-08	2.718E-08	1.046E-08	4.151E-09
2.750	7.763E-07	2.236E-07	8.349E-08	3.643E-08	1.805E-08
2.250	1.654E-06	4.688E-07	2.027E-07	1.153E-07	7.417E-08
1.750	3.058E-06	8.532E-07	4.547E-07	3.077E-07	2.276E-07
1.250	7.367E-06	2.491E-06	1.367E-06	9.440E-07	7.116E-07
0.850	1.652E-05	5.389E-06	2.560E-06	1.588E-06	1.170E-06
0.575	1.006E-05	3.806E-06	2.449E-06	1.908E-06	1.576E-06
0.375	5.521E-06	1.678E-06	1.210E-06	1.023E-06	8.896E-07
0.225	5.878E-06	1.675E-06	1.140E-06	9.370E-07	8.131E-07
0.125	2.262E-06	6.513E-07	3.473E-07	2.416E-07	1.953E-07
0.085	1.136E-06	3.160E-07	2.037E-07	1.742E-07	1.601E-07
0.058	9.464E-07	2.634E-07	1.698E-07	1.451E-07	1.334E-07
0.038	5.679E-07	1.580E-07	1.019E-07	8.709E-08	8.006E-08
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

D.2 Pu-239

Energy (MeV)	12 hours	16 hours	24 hours	36 hours	48 hours
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	5.560E-22	4.906E-22	4.226E-22	3.584E-22	3.052E-22
6.500	1.112E-21	9.812E-22	8.451E-22	7.167E-22	6.105E-22
5.500	1.034E-19	1.949E-20	1.311E-20	1.099E-20	9.222E-21
4.500	1.067E-10	3.782E-11	5.202E-12	2.749E-13	1.456E-14
3.500	1.707E-09	3.229E-10	2.044E-11	5.569E-12	5.569E-12
2.750	1.011E-08	4.354E-09	1.627E-09	7.688E-10	5.793E-10
2.250	4.988E-08	2.384E-08	7.071E-09	3.017E-09	2.453E-09
1.750	1.753E-07	1.111E-07	5.289E-08	2.478E-08	1.705E-08
1.250	5.547E-07	3.525E-07	1.649E-07	7.864E-08	5.360E-08
0.850	9.510E-07	7.186E-07	4.933E-07	3.304E-07	2.455E-07
0.575	1.339E-06	1.023E-06	6.932E-07	4.627E-07	3.382E-07
0.375	7.825E-07	6.213E-07	4.235E-07	2.710E-07	1.903E-07
0.225	7.231E-07	5.917E-07	4.223E-07	2.818E-07	2.053E-07
0.125	1.711E-07	1.444E-07	1.130E-07	8.410E-08	6.657E-08
0.085	1.513E-07	1.398E-07	1.245E-07	1.071E-07	9.369E-08
0.058	1.261E-07	1.165E-07	1.037E-07	8.929E-08	7.807E-08
0.038	7.564E-08	6.990E-08	6.223E-08	5.357E-08	4.684E-08
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	60 hours	72 hours	7 days	14 days	30 days
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	2.590E-22	2.189E-22	5.190E-23	3.538E-24	6.522E-27
6.500	5.179E-22	4.377E-22	1.038E-22	7.077E-24	1.304E-26
5.500	7.731E-21	6.475E-21	1.541E-21	1.194E-22	3.237E-25
4.500	7.720E-16	4.096E-17	1.412E-20	1.154E-21	3.764E-24
3.500	5.737E-12	5.860E-12	5.803E-12	4.363E-12	1.868E-12
2.750	5.295E-10	5.137E-10	4.916E-10	3.758E-10	1.554E-10
2.250	2.245E-09	2.091E-09	1.325E-09	7.304E-10	2.591E-10
1.750	1.453E-08	1.374E-08	1.347E-08	1.039E-08	4.014E-09
1.250	4.307E-08	3.706E-08	2.066E-08	1.210E-08	4.265E-09
0.850	1.959E-07	1.637E-07	6.497E-08	2.128E-08	7.890E-09
0.575	2.609E-07	2.098E-07	7.935E-08	3.487E-08	1.559E-08
0.375	1.424E-07	1.120E-07	4.130E-08	1.922E-08	6.507E-09
0.225	1.595E-07	1.299E-07	4.853E-08	1.712E-08	4.832E-09
0.125	5.520E-08	4.720E-08	1.880E-08	5.487E-09	1.966E-09
0.085	8.312E-08	7.446E-08	3.712E-08	1.431E-08	3.113E-09
0.058	6.926E-08	6.205E-08	3.094E-08	1.192E-08	2.594E-09
0.038	4.156E-08	3.723E-08	1.856E-08	7.153E-09	1.556E-09
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	60 days	90 days	120 days	180 days	1 year
12.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.500	4.652E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00
6.500	9.305E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00
5.500	4.822E-30	7.167E-35	0.000E+00	0.000E+00	0.000E+00
4.500	8.191E-29	1.783E-33	0.000E+00	0.000E+00	0.000E+00
3.500	3.783E-13	8.711E-14	2.960E-14	1.473E-14	1.003E-14
2.750	2.953E-11	6.076E-12	1.357E-12	1.883E-13	9.776E-14
2.250	5.701E-11	1.728E-11	7.787E-12	4.040E-12	2.351E-12
1.750	7.669E-10	1.617E-10	3.347E-11	3.324E-12	1.586E-12
1.250	8.246E-10	1.881E-10	5.163E-11	1.689E-11	1.033E-11
0.850	4.316E-09	3.052E-09	2.306E-09	1.341E-09	2.483E-10
0.575	7.985E-09	5.188E-09	3.580E-09	1.824E-09	3.421E-10
0.375	1.784E-09	8.142E-10	4.470E-10	1.644E-10	3.224E-11
0.225	1.344E-09	5.830E-10	2.970E-10	1.001E-10	2.406E-11
0.125	9.917E-10	5.223E-10	2.835E-10	9.797E-11	2.375E-11
0.085	5.975E-10	2.523E-10	1.445E-10	6.604E-11	2.627E-11
0.058	4.979E-10	2.102E-10	1.204E-10	5.504E-11	2.189E-11
0.038	2.988E-10	1.261E-10	7.226E-11	3.302E-11	1.314E-11
0.025	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Energy (MeV)	2 years	5 years
12.000	0.000E+00	0.000E+00
9.000	0.000E+00	0.000E+00
7.500	0.000E+00	0.000E+00
6.500	0.000E+00	0.000E+00
5.500	0.000E+00	0.000E+00
4.500	0.000E+00	0.000E+00
3.500	5.043E-15	6.417E-16
2.750	4.915E-14	6.245E-15
2.250	1.074E-12	1.080E-13
1.750	7.347E-13	7.903E-14
1.250	5.048E-12	6.293E-13
0.850	3.696E-11	1.970E-11
0.575	7.581E-11	2.482E-11
0.375	1.338E-11	1.786E-12
0.225	9.545E-12	1.009E-12
0.125	9.440E-12	9.693E-13
0.085	1.195E-11	3.123E-12
0.058	9.957E-12	2.603E-12
0.038	5.974E-12	1.562E-12
0.025	0.000E+00	0.000E+00
0.015	0.000E+00	0.000E+00

APPENDIX E

The Transport Matrix is presented. It is a 21 by 21 matrix which will allow any source spectrum and response function to be used to determine a dose rate one meter above that particular ground plane source. Due to the size of the matrix, it will be listed on three separate pages. The first page will contain the left side of the matrix (columns 1-7), the next will contain the middle portion (columns 8-14), while the next page will contain the right side of the matrix (columns 15-21).

Left side of 21 x 21 Transport Matrix. Columns 1-7 and rows 1-21 are presented.

1.0942E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
1.0080E-05	5.4280E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
5.5360E-06	4.8660E-06	2.6200E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
6.5080E-06	4.5700E-06	2.6820E-06	2.6230E-04	0.0000E+00	0.0000E+00	0.0000E+00
7.8080E-06	4.7320E-06	3.0850E-06	3.5110E-06	2.6140E-04	0.0000E+00	0.0000E+00
7.4080E-06	5.6620E-06	3.1110E-06	3.5740E-06	4.5200E-06	2.5500E-04	0.0000E+00
9.9440E-06	6.1920E-06	3.6660E-06	4.3420E-06	5.3000E-06	6.5560E-06	2.5240E-04
6.2220E-06	3.4640E-06	2.0960E-06	2.6370E-06	2.8935E-06	3.4910E-06	4.8840E-06
6.5280E-06	4.4760E-06	2.5530E-06	2.9480E-06	3.2775E-06	3.8325E-06	5.1300E-06
8.5920E-06	5.6830E-06	3.3985E-06	3.6615E-06	3.8105E-06	4.7965E-06	5.8350E-06
1.3978E-05	8.4420E-06	4.7070E-06	5.0600E-06	5.8050E-06	6.6850E-06	7.7600E-06
1.2972E-05	7.7760E-06	4.3170E-06	4.9470E-06	5.4030E-06	5.9790E-06	7.1490E-06
1.1820E-04	4.9665E-05	2.2283E-05	2.0240E-05	1.9130E-05	1.6833E-05	1.5210E-05
5.0910E-05	2.5557E-05	1.2501E-05	1.2528E-05	1.2516E-05	1.3269E-05	1.3932E-05
1.4154E-04	6.7110E-05	3.3015E-05	3.2745E-05	3.3045E-05	3.3195E-05	3.4560E-05
8.9580E-05	4.2850E-05	2.0555E-05	2.0385E-05	2.0455E-05	2.0830E-05	2.1040E-05
8.2884E-05	3.8742E-05	1.8633E-05	1.8852E-05	1.8843E-05	1.8570E-05	1.9050E-05
6.7930E-05	3.2065E-05	1.5450E-05	1.5243E-05	1.5245E-05	1.5230E-05	1.5385E-05
1.3962E-05	6.7050E-06	3.1710E-06	3.1185E-06	3.1680E-06	3.1905E-06	3.2310E-06
8.3440E-07	3.7220E-07	2.0250E-07	1.8180E-07	1.8180E-07	1.8760E-07	1.9800E-07
6.9320E-09	3.8200E-09	7.7240E-10	1.0120E-09	1.2290E-09	1.1120E-09	8.6750E-10

Middle portion of 21 x 21 Transport Matrix. Columns 8-14 and rows 1-21 are presented.

0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
1.2273E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3.5775E-06	1.2168E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3.8175E-06	4.6050E-06	1.1828E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4.6125E-06	5.6225E-06	7.0325E-06	1.1560E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3.9450E-06	4.4085E-06	5.1735E-06	6.9285E-06	6.7491E-05	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
7.0700E-06	6.7538E-06	7.0525E-06	7.8250E-06	6.4365E-06	5.3625E-05	0.0000E+00	0.0000E+00	0.0000E+00
7.0920E-06	7.7775E-06	8.0400E-06	8.6100E-06	5.8050E-06	5.9775E-06	3.2063E-05	0.0000E+00	0.0000E+00
1.8240E-05	1.9230E-05	2.1098E-05	2.4308E-05	1.6871E-05	1.6069E-05	1.1322E-05	0.0000E+00	0.0000E+00
1.0968E-05	1.1550E-05	1.2415E-05	1.3755E-05	9.3030E-06	8.9238E-06	6.2880E-06	0.0000E+00	0.0000E+00
9.7425E-06	1.0292E-05	1.0920E-05	1.1765E-05	7.9677E-06	7.2998E-06	5.1075E-06	0.0000E+00	0.0000E+00
8.0337E-06	8.1750E-06	8.7212E-06	9.5612E-06	6.2528E-06	5.7050E-06	3.8213E-06	0.0000E+00	0.0000E+00
1.6620E-06	1.7108E-06	1.8293E-06	1.9980E-06	1.2600E-06	1.0988E-06	7.8120E-07	0.0000E+00	0.0000E+00
9.4000E-08	9.9900E-08	9.8250E-08	9.5650E-08	7.3680E-08	7.0125E-08	4.0455E-08	0.0000E+00	0.0000E+00
6.1700E-10	5.8400E-10	3.9280E-10	5.4300E-10	7.9980E-10	4.0850E-10	2.7630E-10	0.0000E+00	0.0000E+00

[illegible]

APPENDIX F

The following table shows the mono-energetic Source Conversion Coefficients for each energy group. The photon energy listed is the midpoint energy of each group.

Photon energy (kev)	MSCC (Rad/Hr)/(Mci/km ²)
12000	123.1
9000	95.6
7500	78.8
6500	70.9
5500	63.3
4500	54.1
3500	44.8
2750	37.6
2250	32.3
1750	26.5
1250	20.3
850	15.0
575	10.5
375	7.2
225	4.4
125	2.2
85	1.6
57.5	1.3
37.5	1.7
25	2.7
15	3.4

Appendix G

Source Normalization Constant as a Function of Time

The following table should be used to calculate the dose rate more accurately rather than applying the Way-Wigner approximation to the Source Normalization Constant (K) at one hour.

	K (rad-tissue/Hr)/(kt/km ²)		
time (hr)	U-235	U-238	Pu-239
1.667E-01	5.053E+04	4.941E+04	4.755E+04
3.333E-01	2.643E+04	2.673E+04	2.608E+04
5.000E-01	1.715E+04	1.805E+04	1.754E+04
7.500E-01	1.099E+04	1.171E+04	1.106E+04
1.000E+00	8.064E+03	8.467E+03	7.666E+03
2.000E+00	3.595E+03	3.539E+03	2.870E+03
4.000E+00	1.259E+03	1.235E+03	9.263E+02
6.000E+00	6.449E+02	6.710E+02	4.962E+02
8.000E+00	4.195E+02	4.506E+02	3.410E+02
1.000E+01	3.123E+02	3.353E+02	2.626E+02
1.200E+01	2.496E+02	2.650E+02	2.138E+02
1.600E+01	1.768E+02	1.851E+02	1.544E+02
2.400E+01	1.070E+02	1.138E+02	9.658E+01
3.600E+01	6.333E+01	6.907E+01	6.084E+01
4.800E+01	4.382E+01	4.776E+01	4.435E+01
6.000E+01	3.345E+01	3.616E+01	3.498E+01
7.200E+01	2.725E+01	2.928E+01	2.901E+01
1.680E+02	1.193E+01	1.271E+01	1.270E+01
3.360E+02	5.583E+00	5.736E+00	5.800E+00
7.200E+02	2.258E+00	2.263E+00	2.184E+00
1.440E+03	8.434E-01	8.307E-01	8.070E-01
2.160E+03	4.839E-01	4.590E-01	4.559E-01
2.880E+03	3.375E-01	3.078E-01	3.051E-01
4.320E+03	1.871E-01	1.622E-01	1.596E-01
8.760E+03	3.278E-02	2.959E-02	3.098E-02
1.752E+04	3.668E-03	5.017E-03	6.520E-03
4.380E+04	1.754E-03	3.274E-03	2.305E-03

VI Bibliography

1. Bartine, D. and others, 37 Neutron, 21 Gamma Ray Coupled, P3, Multigroup Library in ANISN Format, DLC-31, Oak Ridge National Laboratory, Oak Ridge, TN, 1979.
2. Bigelow, Winfield S., Photon Transport From a Point Source in the Atmosphere, Unpublished Thesis, Air Force Institute of Technology, Wright-Patterson AFB, OH, 1968.
3. Bridgman, Charles J. Class Notes from NENG 635, The Physics and Effects of Nuclear Explosives, Residual Radiation, School of Engineering, Air Force Institute of Technology, Wright-Patterson AFB, OH, 1989.
4. Cramer, S. N., Applications Guide to the MORSE Monte Carlo Code, ORNL/TM-9355, Oak Ridge National Laboratory, Oak Ridge, TN, 1985.
5. Emmett, M. B., General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code with Combinational Geometry, CCC-203, Oak Ridge National Laboratory, Oak Ridge, TN, July 1984.
6. Glasstone, Samuel and Dolan, Philip J., The Effects of Nuclear Weapons, Washington DC, Government Printing Office, 1977.

7. Kalansky, Gary M., X-Ray Build-Up Factor, Unpublished Thesis, Air Force Institute of Technology, Wright-Patterson AFB, OH, 1978.
8. Mathews, Kirk, Air Force Institute of Technology, Wright-Patterson AFB, OH, private communication, Dec 1989.
9. Millage, Kyle K., Fission Product Decay Characteristics, Unpublished Thesis, Air Force Institute of Technology, Wright-Patterson AFB, OH, 1989.
10. Pace, Joe V., Oak Ridge National Laboratory, Oak Ridge, TN Telephone Interview. October 1989.
11. Walker, William F. and others, Chart of Nuclides (13th edition), General Electric Company, San Jose, CA, 1984.
12. Wood, James, Computational Methods in Reactor Shielding, Pergamon Press, New York, 1982.

REPORT DOCUMENTATION PAGE

1a. REPORT SECURITY CLASSIFICATION <u>Unclassified</u>			1b. RESTRICTIVE MARKINGS	
2a. SECURITY CLASSIFICATION AUTHORITY			3. DISTRIBUTION/AVAILABILITY OF REPORT Approved for public release; distribution unlimited	
2b. DECLASSIFICATION/DOWNGRADING SCHEDULE				
4. PERFORMING ORGANIZATION REPORT NUMBER(S) AFIT/GE/ENP/90M 2			5. MONITORING ORGANIZATION REPORT NUMBER(S)	
6a. NAME OF PERFORMING ORGANIZATION School of Engineering		6b. OFFICE SYMBOL (If applicable) AFIT/ENP	7a. NAME OF MONITORING ORGANIZATION	
6c. ADDRESS (City, State, and ZIP Code) Air Force Institute of Technology (AU) Wright-Patterson AFB, OH 45433-6583			7b. ADDRESS (City, State, and ZIP Code)	
8a. NAME OF FUNDING/SPONSORING ORGANIZATION		8b. OFFICE SYMBOL (If applicable)	9. PROCUREMENT INSTRUMENT IDENTIFICATION NUMBER	
8c. ADDRESS (City, State, and ZIP Code)			10. SOURCE OF FUNDING NUMBERS	
			PROGRAM ELEMENT NO.	PROJECT NO.
			TASK NO.	WORK UNIT ACCESSION NO.
11. TITLE (Include Security Classification) MONTE CARLO DETERMINATION OF GAMMA RAY EXPOSURE FROM A HOMOGENEOUS GROUND PLANE SOURCE				
12. PERSONAL AUTHOR(S) Michael Robert Howard, Captain USAF				
13a. TYPE OF REPORT		13b. TIME COVERED FROM _____ TO _____	14. DATE OF REPORT (Year, Month, Day)	15. PAGE COUNT
16. SUPPLEMENTARY NOTATION				
17. COSATI CODES			18. SUBJECT TERMS (Continue on reverse if necessary and identify by block number)	
FIELD	GROUP	SUB-GROUP	Fallout, Source Normalization Constant,	
24	06			
18	07			
19. ABSTRACT (Continue on reverse if necessary and identify by block number)				
Advisor: Dr. Charles J. Bridgman				
See back of form.				
20. DISTRIBUTION/AVAILABILITY OF ABSTRACT <input checked="" type="checkbox"/> UNCLASSIFIED/UNLIMITED <input type="checkbox"/> SAME AS RPT. <input type="checkbox"/> DTIC USERS			21. ABSTRACT SECURITY CLASSIFICATION <u>Unclassified</u>	
22a. NAME OF RESPONSIBLE INDIVIDUAL Dr Charles J. Bridgman			22b. TELEPHONE (Include Area Code) (513) 255-2012	22c. OFFICE SYMBOL AFIT/ENP

Block 19:

The purpose of this study was to evaluate the gamma ray dose rate from an infinite homogeneous ground plane source of fission products from U-235, U-238, and Pu-239. In this study a Monte Carlo method was used to model the contribution to dose resulting from gamma scatter in the air and in the ground. This was accomplished by using the MORSE code. Two intermediate parameters which were used to determine the dose rate are the Source Conversion Coefficient (SCC) and Source Normalization Constant (K). The mono-energetic SCC was calculated from MORSE and then used to determine the time dependent SCC for a variety of spectra. Both the energy spectra and the gamma emission rates which were used in this study were determined by DKPOWR. The time dependent SCC's and K values are presented for the three fuels. The K values at one hour after burst were 8064 (rads-tissue/hr)/(kt/km²) for U-235 fission products, 8467 (rads-tissue/hr)/(kt/km²) for U-238 fission products, and 7666 (rads-tissue/hr)/(kt/km²) for Pu-239 fission products. The values of K as a function of time was compared to the Way-Wigner approximation, $t^{-1.2}$, and found to differ by as much as 94% within 6 months. In addition to these results, the MORSE code was also used to compute a transport matrix which will permit the use of any other source spectrum and any other response function to be converted to a ground dose rate. This transport matrix is also presented.

Handwritten notes:
X
19
fig. 6
10
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100